John D. O'Toole
Vice President

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November 29, 1982

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Dear Mr. Varga:

Transmitted as Attachment A to this letter is our response to your letter dated October 4, 1982 requesting additional information concerning "PWR Main Steam Line Break with Continued Feedwater Addition, IE Bulletin No. 80-04."

Should you or your staff have any questions, please contact us.

Very truly yours,

John D. O'Toole

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ATTACHMENT A

Response to NRC's October 4, 1982
request for additional information concerning
"PWR Main Steam Line Break with Continued
Feedwater Addition, IE Bulletin No. 80-04"

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
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Question A1

An evaluation of the potential for a single active failure in th MFW system which could cause the greatest feedwater flow to the affected steam generator during a MSLB accident and a determination of MFW flow rate to the affected generator if a single active failure were to occur.

Response:

A main steam line break will initiate a safety injection signal via the steam line break protection circuitry. The safety injection signal will isolate main feedwater by automatically closing the air-operated main and low flow bypass feedwater regulating valves and closing both motor operated main boiler feedwater pump discharge valves, thus redundantly isolating both main boiler feedwater pumps. In addition, if offsite power is available, closure of the series motorized block valves associated with each main and low flow bypass feedwater regulating valves will also receive an automatic closure signal.

Steam line break protection logic through the safety injection signal to the valves is redundant and safety grade. Accordingly no single failure will preclude main feedwater isolation under a design basis steam line break event.

To further assure that FSAR analyses concerning both containment pressure and core reactivity response during a main steam line break event have been conservatively determined, an analysis assuming failure of a fast acting main feedwater regulating valve will be performed. Continued main feedwater flow will be assumed for the additional time required for the motor operated valve to close. The additional feedwater flow to the affected steam generator will be calculated and its effect on containment pressure and core reactivity responses determined. Our current schedule for completion of this work is July,1983. Should the results of this analysis indicate a potential for exceeding containment design pressure or a change in core reactivity exceeding the applicable acceptance criteria, a response to questions A.2 and B.1 of your request will be provided along with a schedule for implementation of corrective action.

Question A.3. In the submittal dated May 8, 1980, it was stated that emergency procedures for the MSLB required identification of the affected steam generator and isolation of the AFW flow to the generator. Explain what is needed to be done to do this including operator action and the times assumed for these actions. Provide justification for you assumptions.

Response: Following a containment pressurization from a steam line break, the operator first checks for high containment radiation to determine that it is not a main coolant break. He then checks the four steam generators for level and pressure to determine the broken steam line. The operator then isolates auxiliary feedwater to the affected steam generator. Simulator training indicates that identification of the affected steam generator will normally be accomplished within one (1) minute of accident initiation. Isolation of AFW flow to the affected steam generator follows immediately. Training experience further indicates that both identification of the affected steam generator and isolation of AFW flow should always be accomplished in less than ten minutes.